ACCESSION #: 9111010191
LICENSEE EVENT REPORT (LER)

FACILITY NAME: DIABLO CANYON UNIT 1 PAGE: 1 OF 14

DOCKET NUMBER: 05000275

TITLE: REACTOR TRIP RESULTING FROM FAILED OPEN PRESSURIZER SPRAY VALVE
DUE TO INCORRECT SCREW INSTALLATION
EVENT DATE: 12/24/90 LER #: 90-017-01 REPORT DATE: 10/23/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 088

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)
50.73(a)(2)(i)(B)

LICENSEE CONTACT FOR THIS LER:
NAME: MARTIN T. HUG, SENIOR REGULATORY COMPLIANCE ENGINEER TELEPHONE: (805) 545-4005

COMPONENT FAILURE DESCRIPTION:
CAUSE: E SYSTEM: SB COMPONENT: V MANUFACTURER: C635
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On December 24, 1990, at 0318 PST, with Unit 1 in Mode 1 (Power operation) at 88 percent power, a reactor trip and safety injection occurred due to low pressurizer pressure. During the recovery from the trip, reactor coolant system (RCS) cooldown exceeded the allowable rate of 100 degrees Fahrenheit per hour of Technical Specification 3.4.9. An Unusual Event was declared at 0320 PST. A one-hour emergency report required by 10 CFR 50.72 (a)(1)(i) was made on December 24, 1990, at 0342 PST. On December 24, 1990, an Event Investigation Team was formed to investigate the event.

The cause of the trip was a pressurizer spray valve that failed open due to its feedback linkage becoming disconnected because the locking device for the screw holding the linkage to the valve stem had not been

installed. The major contributor to the overcooling of the RCS was the failure of a condenser steam dump valve pilot stem due to sticking, which resulted in bending of the main stem and thus holding open the valve.

Corrective actions for the event include revising Maintenance Procedure I-2.25-1 to address the use of appropriate locking devices on feedback linkages, revising Abnormal Operating Procedure AP-13 to provide additional guidance for dealing with failed open pressurizer spray valves, revising Emergency Procedure E-0 to provide guidance on closing the main steam isolation valves during cooldown transients, revising design drawings to clarify the installation of the feedback linkage, sending a letter to the vendor describing the feedback linkage problem and asking them to review the documentation used for the linkage assembly for adequacy. Also, the seating angles of the condenser steam dump valves were modified to alleviate sticking.

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END OF ABSTRACT

I. Plant Conditions

Unit 1 was in Mode 1 (Power Operation) at 88 percent power.

II. Description of Event

A. Event:

On December 24, 1990, a power increase was in progress following condenser cleaning. At 0316 PST, the control operator noted pressurizer (AB)(PZR) pressure decreasing rapidly, and notified the shift foreman.

The cause of the decreasing pressure was diagnosed as a failed open pressurizer spray valve, since it was noted that pressurizer spray valve (AB)(PZR)(PCV) RCS-1-PCV-455B did not indicate closed with zero demand on the valve controller. It was apparent to the operators that they could not correct the problem.

On December 24, 1990, at 0318 PST, with the unit at 88 percent power, operators started to initiate a manual reactor trip. However, an automatic reactor trip and safety injection (JE) (SI) due to low pressurizer pressure occurred prior to the manual trip. On December 24, 1990, at 0320 PST, an Unusual Event (UE) was declared due to the SI. A one-hour emergency report was made in accordance with 10 CFR 50.72(a)(1)(i) at 0347 PST.

The control room operators entered Emergency Procedure (EP) E-0, "Reactor Trip or Safety Injection," and verified that the trip and SI were properly initiated, and that other plant parameters were as expected. Since instrument air (LD) had been isolated from containment due to the containment isolation (JM) signal generated by the SI signal, the pressurizer spray valve had begun to close, though pressure had not yet begun to recover. At 0323 PST, Reactor Coolant Pump (AB)(P) (RCP) 1-2 was secured since one of the EP E-0 evaluations verifies the position of pressurizer spray valves and instructs the operators to secure the associated RCP if a valve is stuck open and reactor coolant system (AB) (RCS) pressure is decreasing.

The diagnostic section in EP E-0 identified no other anticipated problems except for a continuing cooldown. Per EP E-0, the operators verified that the condenser steam dump valves (SB) (V) (SDVs) were closed and reduced auxiliary feedwater (BA) (AFW) flow to the minimum required amount. All condenser SDVs indicated closed, even though a subsequent review determined this was not actually the case. One of the condenser SDVs indicated closed because the main stem was in the closed position; however, investigation identified that the valve plug was not attached to the main stem, and the pilot valve was actually open.

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Once all SI termination criteria of EP E-0 were met, the operators transitioned from procedure EP E-0 to EP E-1.1, "SI Termination." EP E-1.1 directed the operators to reset the SI and containment isolation signals, and re-establish instrument air pressure to containment. These steps are performed to allow normal RCS charging (BQ) flow and letdown flow to be established. With normal instrument air pressure to containment, RCS-1-PCV-455B re-opened at 0330 PST. Because of the differential pressure conditions existing with RCP 1-2 shut down, pressurizer spray flow through RCS-1-PCV-455B was re-initiated due to motive flow from the other RCPs. RCS pressure began to decrease. EP E-1.1 evaluates RCS pressure, and, if it is noted to be decreasing, directs the operators to go to EP E-1, "Loss of Reactor or Secondary Coolant."

During the performance of EP E-1, the shift foreman concluded that some pressurizer spray might be continuing. At 0339 PST, the shift foreman ordered that RCP 1-1 be shutdown to terminate pressurizer spray. This terminated the RCS pressure decrease and allowed the operators to return to EP E-1.1. At 0342 PST, it was noted that the RCS cooldown rate of approximately 101 degrees F in 24 minutes experienced during this event exceeded the maximum allowable cooldown rate for the RCS of 100 degrees F in any hour as specified in Technical Specification (TS) 3.4.9.1.

Since an RCS cooldown was still in progress, and control of pressurizer pressure and level had been restored, the shift foreman directed that the main steam isolation valves (SB)(ISV) (MSIVs) be closed. The unit was stabilized in Mode 3 (Hot Standby). The UE was terminated on December 24, 1990, at 0500 PST.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

1. The feedback linkage for pressurizer spray valve RCS-1-PCV-455B became disconnected because of a missing elastic stop nut, causing the valve to fail open.

2. Condenser SDV PCV-1 stuck partially open due to a broken pilot valve stem and resulting bent valve main stem.

C. Dates and Approximate Times for Major Occurrences:

1. December 24, 1990, at 0316 PST: Pressurizer pressure low alarm.

2. December 24, 1990, at 0318 PST: Event/Discovery date - Automatic reactor trip and SI on low RCS/pressurizer pressure.

3. December 24, 1990, at 0320 PST: UE declared.

4. December 24, 1990, at 0342 PST: Event/Discovery date - RCS cool down rate exceeded TS 3.4.9 allowable rate.

5. December 24, 1990, at 0347 PST: One-hour emergency notification made in accordance with 10 CFR 50.72(a)(1)(i).

6. December 24, 1990, at 0500 PST: UE terminated with unit stable in Mode 3.

D. Other Systems or Secondary Functions Affected:

1. One SDV stuck partially open due to failure of the pilot valve stem and resulting bent valve main stem.

2. The main annunciator typewriter (IB)(TPW) registered all valid alarms. The typewriter also registered several inappropriate alarms. Though these alarms were unwarranted, they did not affect the response to the transient.

3. Feedwater heater relief valve (SJ)(RV) FW-1-RV-37 stuck open during the event. The valve was disassembled and large amounts of corrosion products were found. This condition created binding on internal parts and also would cause debris to settle on the seats. The valve was cleaned, rebuilt, and tested satisfactorily.

E. Method of Discovery:

The event was immediately apparent to licensed plant operators due to alarms and indications received in the control room.

F. Operators Actions:

Operators took the appropriate actions to stabilize the plant in Mode 3. Operators decided not to close the MSIVs at the beginning of the transient since closure of the MSIVs after past reactor trips and SIs had caused rapid heat-up and swell of the RCS, and could potentially result in overpressurization of the RCS.

G. Safety System Responses:

1. The reactor trip breakers opened (AA)(BKR).

2. The control rod drive mechanism (AA)(DRIV) allowed the control rods to drop into the core.

3. An SI signal was initiated on low pressurizer pressure.

4. The diesel generators (EK)(DG) started.

5. The SI pumps (BQ)(P) started.

6. The residual heat removal pumps (BP) (P) started.

7. The charging pumps (BQ)(P) started.

8. The motor-driven AFW pumps (BA)(MO)(P) started as designed.

III. Cause of the Event

A. Immediate Cause:

Reactor Trip and SI:

1. The immediate cause of the reactor trip was low pressurizer pressure due to pressurizer spray valve RCS-1-PCV-455B failing open.

Overcooling:

2. Simulation:

To fully investigate the plant response following the December 24, 1990, reactor trip and SI, and to determine the cause for the rapid cooldown, the event was recreated on the Diablo Canyon Power Plant (DCPP) simulator. The initial conditions as described in the control room logs for December 24, 1990, were used as the initial conditions for the simulator. Two separate simulations were performed. In one simulation, pressurizer spray valve RCS-1-PCV-455B was failed open and condenser SDV PCV-1 was not failed open. In the other simulation, the spray valve and the condenser SDV were both failed open. The times to perform operator actions were taken from the annunciator printout for the actual trip.

The simulations identified that the open failure of one condenser SDV causes the maximum cooldown rate specified in TS to be exceeded. The actual plant cooldown was not, however, as severe as the simulator cooldown.

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Event:

The cooldown proceeded normally from the time of the reactor trip with RCS Tavg temperature at approximately 575 degrees F, to the point when the condenser SDVs were designed to close, at 547 degrees F. The cooldown should have terminated at approximately 520 degrees F, as the SDVs should have closed and the plant reached thermal equilibrium. The RCS normally continues to cool below 547 degrees F after the SDVs are closed due to AFW flow into the steam generators, and steam leakage in the secondary plant through the main steam reheaters and the main turbine drains.

During this event, the RCS continued to cool after reaching 520 degrees F. The overcooling of the RCS after the reactor tripping can be attributed to several coincident factors. Coincident with the trip, an SI was initiated. Cooler water was injected into the RCS, reducing the RCS temperature. To eliminate pressurizer spray flow, two RCPs were shut down, removing their heat input to the RCS. The unit was ramping up from 50 percent power, when it tripped at 88 percent power. The unit had been at 50 percent power for approximately two days for condenser cleaning. As a result of operating at a reduced power level, the decay heat input to the RCS was not as great as the decay heat after a trip from full power.
Heat removal from the secondary plant to condenser was greater than normal due to the failure of the pilot valve stem in PCV-1, which allowed the steam flow to bend the

valve main stem and caused the main plug to be held partially open. This contributed to the increased cooldown rate. The partially held open main plug may also have contributed to the differences observed between simulator and actual plant behavior.

The immediate cause of the overcooling was a combination of all the above factors; however, the most significant contributor was the failure of PCV-1.

B. Root Cause:

1. Reactor Trip:

A containment entry was made on December 24, 1990, by the Instrumentation and Controls (I&C) engineers to determine the as-found condition of RCS-1-PCV-455B. When the valve was examined, it was determined that the valve had failed open due to the positioner feedback linkage coming loose from the valve stem attachment plate. When the feedback linkage came loose, the feedback arm and the cam internal to the positioner moved from its demanded position to a point where positioner spring tension stopped the cam. When the cam assumed this position,

output from the positioner caused RCS-1-PCV-455B to start to open. With no feedback available on valve position, the valve continued to open until it reached its full open position.

The examination of the as-found condition of the feedback linkage further determined that the feedback linkage had come loose because the machine screw which connects the feedback linkage to the valve stem attachment plate had become disconnected. The I&C department concluded that during normal operation, the torque applied to the machine screw can cause the machine screw to come loose from the valve attachment plate if an elastic stop nut or other locking device is not used to prevent it from coming loose.

A review of the manufacturer's design drawings for

RCS-1-PCV-455B indicated an elastic stop nut in the bill of material for the positioner. However, the drawing did not provide a detail showing the installation of the elastic stop nut.

Further investigation identified that this valve was procured as an assembly with the actuator and positioner attached to the valve at the time of procurement. The elastic stop nut presumably would have been furnished as part of this assembly.

A review of all DCPP work orders performed on this valve since original procurement identified no activities which would require disconnection of the feedback linkage from the valve stem attachment plate.

The root cause for the failure of this valve can be attributed to two possible causes.

a. Improper assembly by the vendor prior to receipt of the positioner.

b. Evidence indicates that the feedback linkage was not adjusted or removed by DCPP personnel; however, it is possible that the linkage was removed during valve installation and set-up, and reassembled incorrectly.

2. Failure of PCV-1 contributing to overcooling:

The following investigative actions were taken to identify the root cause of the valve failure:

a. The as-found condition of the valve was identified. The main stem was bent, and the threaded connection between the pilot stem and the main stem was broken.

b. A metallurgical analysis of the broken pilot valve stem connection was performed. The analysis concluded that the pilot stem failed due to a ductile overload, and not as a result of fatigue.

c. The vendor was contacted to identify any other similar failures, and the vendor also inspected PCV-1 and the pilot valve onsite. The vendor was not aware of any similar failures, but did state that if the connection between the pilot stem and the main stem were not correct, or if the main stem were bent, a failure of the connection could result. The vendor also stated that the pilot stem and the main stem should not be assembled in the field because it is difficult to connect them correctly.

d. A maintenance history search was performed to identify past maintenance that may have affected the operability of the valve. The search identified that the pilot stem and main stem had been reassembled in the field in July 1990. Preliminary results of investigation indicated that pilot valve stem failure was due to incorrect assembly of the PCV-1 main stem to the pilot valve stem by PG&E. However, several subsequent failures of manufacturer-assembled SDVs at DCPP have shown this not to be the case.

e. A multi-disciplinary review team was formed to review this and other subsequent DCPP SDV pilot stem failures in greater detail. A program of diagnostic testing was performed to validate the root cause and proposed corrective actions.

The review team determined that the pilot valve stem failure was the result of two factors. The first factor was sticking of the main valve plug to its seat due to microwelding. As provided by the manufacturer, the valve plug has a dual angle seat design. This design allows for a thin line seating surface contact with the valve seat. During normal power operation, the valve is closed with pressure-assisted sealing due to main steam pressure behind the plug. This large pressure force, concentrated on the thin circular seating surface, caused the plug to wedge and microweld to the valve seat. The seating surfaces of the SDVs exhibiting this type of failure had been replaced with new or re-machined components in 1990, which provided the clean and sharp seating surfaces that are believed to aggravate microwelding. With the valve sticking to its seat, attempts to open the valve resulted in the actuator imparting a large compressive force to the valve stem in an attempt to break the plug away from its seat. When the plug finally broke free, the plug was subjected to a large acceleration.

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The second factor was the installation of mechanical stops within the valve actuator in 1989-1990. As the valve stem travels over the range of its stroke, the stem is stopped by the mechanical stops. The main valve plug was rapidly decelerated when it struck the now motionless pilot plug. This rapid deceleration imparted a large tensile stress to the pilot valve stem. This stress exceeded the ultimate tensile stress of the pilot valve stem material.

The review team considered this root cause to be a result of an inadequate design of the trim components by the manufacturer. The seat angles on the main plug create the potential for microwelding of the plug to the main seat. The actuator has the ability to break the plug away from the seat, but in doing so imparts a large acceleration to the plug. The cross-sectional diameter of the pilot plug stem is not sufficient to withstand the impact of the main plug, thus resulting in stem failure. The review team considered the sticking of the plug to the seat to be a situation that should have been anticipated by the manufacturer and that the trim components should have been sized to withstand any dynamic stresses resulting from this sticking.

Manufacturing deficiencies resulting in poor thread engagement of the pilot plug to the main stem may have been a contributor to the stem failures, although the mechanism that initiated the failures was still considered to be the microwelding-and-breakaway scenario described above. The installation of hard travel stops in the actuator was also a contributor to the failed stems. These stops had been installed to address the existing problem of breakage of positioner cam hubs due to stem overtravel that had also been occurring as a result of the microwelding-and-breakaway scenario.

IV. Analysis of the Event

1. Reactor Trip:

Accidental depressurization of the RCS is identified in the Final Safety Analysis Report (FSAR) Update as a Condition II event - Faults of Moderate Frequency. The effects of this event are analyzed in Section 15.2.12. The results of this analysis assure the minimum departure from nucleate boiling ratio (DNBR) remains in excess of 1.30 for this event. A failed open pressurizer spray valve and subsequent depressurization of the RCS is an analyzed condition in Chapter 5 of the FSAR Update.

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2. Overcooling:

A Westinghouse engineering evaluation of the RCS considered the impact of thermal transient upon the pressurizer, reactor vessel, RCS piping, the thick metal of the steam generators, and the RCPs.

a. Pressurizer

The pressurizer thermal transient limits are specified in TS 3.4.9.2, "Pressurizer." The 200 degrees F per hour limiting condition for operation limit in the TS was not exceeded by this event. Consequently there has been no significant adverse effect on this component.

b. RCS Other Than the Pressurizer

The DCPP Unit 1 rapid cooldown transient of greater than 100 degrees F in any hour was evaluated by reviewing temperature and pressure data and comparing these with evaluations of similar transients at other plants. The comparison showed that the DCPP Unit 1 rapid cooldown event is bounded by transients previously analyzed for other plants. It was concluded from the Westinghouse evaluation that the above described DCPP Unit 1 transient did not adversely affect the structural integrity of the affected components and system, and that the RCS could be returned to normal operating temperature and pressure and the unit restarted safely.

3. 40 Percent SDVs:

A failure of one or more 40 percent SDVs, concurrent with a failure of an MSIV to close, creates a cooldown which is less severe than the licensing basis double-ended rupture in the steamline which has previously been analyzed. However, the effect of SDV and MSIV failure has not previously been considered in conjunction with all FSAR Update Chapter 15 accident analyses. As the root cause for the SDV failure is common for all 40 percent SDVs at DCPP, the effect on safety has been evaluated for Unit 1 and for Unit 2 assuming that either unit could experience the failure of all 12 40 percent SDVs to reclose after receiving an open signal.

Two bounding scenarios were postulated:

a. Steamline Rupture

The first scenario results in one steam generator blowing down through a steam line break upstream of the flow limiting device (the FSAR Update Chapter 15 analysis basis), and additional

release of steam from a second steam generator through the SDVs to the condenser due to MSIV failure in an unfaulted steam generator line. This scenario is a bounding case with respect to reactor core response.

For Unit 1, the actual core conditions for the first 75 effective full-power days (EFPD) of Unit 1 fuel cycle 5 were such that the resulting transient would be bounded by the FSAR Update analysis of the main steamline rupture, which is based on end-of-core-life conditions. The prior analysis remained limiting mainly because there is less reactivity feedback early in core life. Corrective actions to prevent recurrence of SDV failures were completed prior to 75 EFPD of fuel burnup.

For Unit 2, the limiting steamline rupture case was re-analyzed to take credit for the steam line check valves which are not modelled in the FSAR Update analysis (justified since the steam line check valves are tested in accordance with the DCPP Inservice Testing Program), and to model a higher reactivity coefficient for the boron from the accumulators and the Refueling Water Storage Tank. The results of this analysis show that the DNBR design basis was met and thus core integrity is demonstrated.

b. Steam Generator Tube Rupture

The second scenario results in a radioactive water release to the environment due to a steam generator tube rupture concurrent with a failure of one or more 40 percent SDV and an MSIV. This scenario is a bounding case with respect to offsite dose releases.

The calculated 2-hour exclusion area boundary (EAB) thyroid dose for this scenario equals approximately 0.14 rem assuming:

o maximum design flow for a double-ended failure of a single steam generator tube,

o the entire primary to secondary break flow is released (no iodine partitioning),

o no post-trip iodine spike occurs,

o the actual measured I-131 Dose Equivalent of Unit 2, and

o either the MSIV or SDV is isolated within one hour.

The calculated EAB thyroid dose of approximately 0.14 rem is a factor of about 210 less than the design basis steam generator tube rupture 2-hour EAB dose of 28.8 rem. Therefore, the prior analysis remains limiting.

4. Summary

The accidental depressurization of DCPP Unit 1 is an analyzed accident as described above, and the thermal/pressure transient effects on the RCS have been evaluated. Based on the above information, the response of the reactor protective devices and the thermal/pressure transient effects on the RCS and components has been evaluated. The accidental depressurization event has been shown to not have had an adverse impact on the core or the RCS components. The thermal/pressure transient effects on the RCS were shown to not have jeopardized the integrity of the RCS pressure boundary. The potential failure of one or more 40 percent SDVs has been analyzed and determined not to affect the conclusions of prior analyses. Consequently, the accidental RCS depressurization and thermal/pressure transient did not adversely affect the health and safety of the public.

V. Corrective Actions

A. Immediate Corrective Actions:

1. The plant was stabilized in Mode 3.

2. An Event Investigation team and an action plan were formed to investigate the causes and corrective actions for the event and the associated problems.

3. The feedback linkage for RCS-1-PCV-455B in Unit 1 was properly reassembled. A locknut was installed to assure that the machine screw connecting the feedback linkage to the valve is properly secured.

4. The other 3 spray valves were inspected. RCS-1-PCV-455A, RCS- 2-PCV-455A and RCS-2-PCV-455B were found to be assembled with elastic stop nuts.

5. A selected sample of Units 1 and 2 valve feedback linkage arms, which included those on all the feedwater regulating valves, all the feedwater regulating bypass valves, and all the main steam dumps was examined to assure that the linkages had adequate locking devices installed. No problems were identified with these valve feedback linkages.

6. The main stem to pilot valve stem assembly of PCV-1 was replaced, and the valve was tested and returned to service.

7. An INPO Network entry was made regarding the DCPP SDV events.

B. Corrective Actions to Prevent Recurrence:

Reactor Trip:

1. Maintenance Procedure (MP) I-2.25-1, "Control Valve Travel and Bench Set Adjustments," was revised to address the use of an appropriate locking device when installing feedback linkage arms.

2. The appropriate loop tests were revised to reference the revision to MP I-2.25-1.

3. A design drawing change was issued to clarify the installation of the feedback linkages.

4. A Maintenance Bulletin was issued discussing this event and the lack of adequate locking devices on feedback linkages.

5. A letter was sent to the vendor describing the problem with the feedback linkage, and requesting the vendor to review the adequacy of the documentation used for valve to positioner assembly.

Overcooling:

6. Operating Procedure (OP) AP-13, "Malfunction of Reactor Pressure Control System," was revised to provide additional guidance for dealing with a failed open pressurizer spray valve.

7. EP E-09 "Reactor Trip or Safety Injection," was revised to provide additional guidance concerning operation of the MSIVs during cooldown transients.

SDV PCV-1:

8. All Unit 1 and Unit 2 40 percent steam dump valve trim kits were replaced. This work included modifying the seating surfaces with a new seating angle designed to alleviate valve sticking, increasing the pilot stem diameter, and changing the material of several valve components to a material more resistant to microwelding.

Main annunciator typewriter:

9. An investigation determined that in addition to this event, spurious alarms have occurred previously during transients when electrical switching has occurred. To alleviate this, the

annunciator input cards were replaced with newer versions that are less susceptible to noise.

VI. Additional Information

A. Failed Components:

The pilot valve stem for PCV-1 fractured.

B. Previous LERs:

None.

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James D. Shiffer
Senior Vice President and
General Manager
Nuclear Power Generation

October 23, 1991

PG&E Letter No. DCL-91-253

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 1-90-017-01
Reactor Trip Resulting from Failed Open Pressurizer Spray Valve
Due to Incorrect Screw Installation

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(iv) and 10 CFR 50.73 (a)(2)(i)(B), PG&E is submitting the enclosed revised Licensee Event Report (LER) concerning a reactor trip resulting from a failed pressurizer spray valve. The pressurizer spray valve failed open due to its feedback linkage becoming disconnected because a locking device had not been installed. During the recovery from the reactor trip, the reactor coolant system was cooled at a rate greater than 100 degrees Fahrenheit in any hour, in violation of Technical Specifications. This revision is being submitted to present the final results of investigations and corrective actions regarding the overcooling.

This event has in no way affected the health and safety of the public.

Sincerely,

James D. Shiffer

cc: Ann P. Hodgdon
John B. Martin
Phillip J. Morrill
Harry Rood
Howard J. Wong
CPUC
Diablo Distribution

DC1-90-TI-N090
DC0-90-TI-N091

Enclosure

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